September 15, 1989

Docket No. 50-271

See Correction letter of
[illegible]

Mr. L. A. Tremblay
Licensing Engineering
Vermont Yankee Nuclear Power Corporation
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Tremblay:

SUBJECT: ISSUANCE OF AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-28 - VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. 73363)

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment is in response to your application dated May 12, 1989. Clarifying information was provided July 14, 1989.

This amendment modifies the Technical Specifications to eliminate cycle-specific parameter limits, and instead refer to a Core Operating Limits Report for the value of these parameters.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

This completes action under TAC 73363.

Sincerely,

Original Signed By

Morton B. Fairtile, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 116 to License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

DF01

[TAC 73363]

PDI-3/PM	PDI-3/LA	OGC	PDI-3/D
MBFairtile:rw	MRushbrook		RHWessman
9/12/89	8/25/89	9/16/89	9/14/89

MBF 9/14/89

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Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated May 12, 1989 as supplemented on July 14, 1989 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 116, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 15, 1989

PDI-3/PM	PDI-3/LA	OGC	PDI-3/D
MBFairtile:rw	~~MRushbrook~~		RHWessman
8/30/89	8/25/89	9/6/89	9/14/89
MBF 9/14/89			

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ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2	2
4b	4b
5-b	5-b
6-a	6-a
14-a	14-a
48	48
110	110
180a	180a
180b	180b
180-c	180-c
180-d	180-d
180f	180f
180-g	180-g
180-h	180-h
180-l through 180-n8	180-l
180-0	-
180-01	-
209	209
-	209a

G. Instrument Functional Test - An instrument functional test shall be:

1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

2. Bistable channels - the injection of a signal into the sensor to verify the operability including alarm and/or trip functions.

H. Log System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

I. Minimum Critical Power Ratio - The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the appropriate NRC-approved critical power correlation to the actual assembly operating power.

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

K. Operable - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

L. Operating - Operating means that a system or component is performing its intended functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

JJ. Process Control Program (PCP) - A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

KK. Gaseous Radwaste Treatment System - The Augmented Off-Gas System (AOG) is the gaseous radwaste treatment system which has been designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

LL. Ventilation Exhaust Treatment System - The Radwaste Building and AOG Building ventilation HEPA filters are ventilation exhaust treatment systems which have been designed and installed to reduce radioactive material in particulate form in gaseous effluents by passing ventilation air through HEPA filters for the purpose of removing radioactive particulates from the gaseous exhaust stream prior to release to the environment. Engineered safety feature atmospheric cleanup systems, such as the Standby Gas Treatment (SBGT) System, are not considered to be ventilation exhaust treatment system components.

MM. Vent/Purging - Vent/Purging is the controlled process of discharging air or gas from the primary containment to control temperature, pressure, humidity, concentration or other operating conditions.

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.4. Plant operation within these operating limits is addressed in individual specifications.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is defined in the Core Operating Limits Report.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

APRM gain shall be increased by the ratio: $\frac{\text{MFLPD}}{\text{FRP}}$

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is defined in the Core Operating Limits Report.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions. For single recirculation loop operation, the APRM flux scram trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR defined in the Core Operating Limits Report.

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. Refer to the Core Operating Limits Report for acceptable values for N. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period.

7. The trip may be bypassed when the reactor power is $\leq$30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

8. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

9. With one RBM channel inoperable:

   a. Verify that the reactor is not operating on a limiting control rod pattern, and

   b. Restore the inoperable RBM channel to operable status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

3.6 LIMITING CONDITION FOR OPERATION

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

   b. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow between 34% and 45% of rated (Region 2 of Figure 3.6.4):

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle.

G. Single Loop Operation

1. With one recirculation pump not in operation, core flow between 34% and 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Region 2), establish baseline APRM and LPRM(1) neutron flux noise levels prior to entering this region, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.6.4.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values provided in the Core Operating Limits Report. For single recirculation loop operation, the limiting values shall be the values provided in the Core Operating Limits Report listed under the heading "Single Loop Operation." If at any time during steady-state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

limits. If the APLHGR is not returned to within prescribed limits within two (2) hours, the reactor shall be brought to the shutdown conditions within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

B. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR provided in the Core Operating Limits Report.

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

C. Minimum Critical Power Ratio

MCPR shall be determined daily during reactor power operation at $\geq$25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6.

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

1. During steady state power operation the MCPR operating value shall be equal to or greater than the MCPR limits provided in the Core Operating Limits Report. For single recirculation loop operation, the MCPR Limits at rated flow are also provided in the Core Operating Limits Report. For core flows other than rated, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is provided in the Core Operating Limits Report. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be brought to shutdown condition, within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

Bases:

3.11 Fuel Rods

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

(Note: All exposure increments in this Technical Specification section are expressed in terms of megawatt-days per short ton.)

The MAPLHGR reduction factor of 0.83 for single recirculation loop operation is based on the assumption that the coastdown flow from the unbroken recirculation loop would not be available during a postulated large break in the active recirculation loop, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation." February 1983.

This page has been deleted.

Bases:

3.11B Linear Heat Generation Rate (LHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

Bases:

3.11C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

1. The MCPR operating limit is a cycle-dependent parameter which can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the Fuel Cladding Integrity Safety Limit (FCISL) for potential abnormal occurrences. The MCPR operating limits are justified by the analyses, the results of which are presented in the current cycle's Core Performance Analysis Report. Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analysis methods. The 0.01 increase in MCPR operating limits for single loop operation accounts for increased core flow measurement and TIP reading uncertainties, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation," February 1983.

Pages 180-1 through 180-01 have been deleted.

The next page is 180p.

2. Annual Report

An annual report covering the previous calendar year shall be submitted prior to March 1 of each year. The annual report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurement. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Statistical Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the fifteenth of each month following the calendar month covered by the report. These reports shall include a narrative summary of operating experience during the report period which describes the operation of the facility.

4. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: (a) The Average Planar Linear Heat Generation Ratio (APLHGR) for Specification 3.11.A and 3.6.G.1a, (b) The K_f core flow adjustment factor for Specification 3.11.C., and (c) The Minimum Critical Power Ratio (MCPR) for Specification 3.11.C and 3.6.G.1a. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).

1/ This tabulation supplements the requirements of 20.407 of 10CFR Part 20.

Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code Qualification and Application," YAEC-1265P, June 1981 (Approved by NRC SER, dated September 27, 1985).

Report, A. A. F. Ansari and J. T. Cronin, "Methods for the Analysis of Boiling Water Reactors: A System Transient Analysis Model (RETRAN)," YAEC-1233, April 1981. (Approved by NRC SERs, dated November 27, 1981 and September 4, 1984).

Report, A. A. F. Ansari, K. J. Burns and D. K. Beller, "Methods for the Analysis of Boiling Water Reactors: Transient Critical Power Ratio Analysis (RETRAN-TCPYA01)," YAEC-1299P, March 1982 (Approved by NRC SER, dated September 15, 1982).

Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (Approved by NRC SER, dated November 30, 1977).

Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A-9, GE Company Proprietary, September 1988, as amended.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. <u>Reportable Occurrences</u>

This section deleted.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 116 TO LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated May 12, 1989 (Ref. 1), as clarified by letter dated July 14, 1989 (Ref. 2), Vermont Yankee Nuclear Power Corporation (the licensee) proposed changes to the Technical Specifications (TS) for Vermont Yankee. The clarification of July 14, 1989 did not affect the staff's proposed no significant hazards consideration determination. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 3).

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the Core Operation Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specifications 3.6.G.Ia and 3.11.A

The average planar linear heat generation rate (APLHGR) limits for these specifications are provided in the COLR.

(b) Specification 3.11.C

The K_f factors that are applied to the operating limit minimum critical power ratio (MCPR) for this specification are provided in the COLR.

(c) Specifications 3.6.G.Ia and 3.11.C

The minimum critical power ratio (MCPR) limits for these specifications are provided in the COLR.

(d) Specifications 2.1.A.Ia, 2.1.B.1, and 3.11.B

The linear heat generation rate (LHGR) limits for these specifications are provided in the COLR.

(e) Specification 3.2 - Protective Instrument Systems (Control Rod Block Instrumentation - Table 3.2.5)

The value of N used in the Rod Block Monitor Setpoint and specified in Note 5 to Table 3.2.5 is provided in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. We conclude that the changes to these bases are acceptable.

(3) Specification 6.7.A.4 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodology and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (approved by NRC SER, dated September 15, 1982).

(b) Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (approved by NRC SER, dated September 15, 1982).

(c) Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (approved by NRC SER, dated September 15, 1982).

(d) Report, S. P. Schultz and K. E. St. John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1242P, April 1981 (approved by NRC SER, dated September 27, 1985).

(e) Report, S. P. Schultz and K. E. St. John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code Qualification and Application," YAEC-1265P, June 1981 (approved by NRC SER, dated September 27, 1985).

(f) Report, A. A. F. Ansari and J. T. Cronin, "Methods for the Analysis of Boiling Water Reactors: A System Transient Analysis Model (RETRAN)," YAEC-1233, April 1981 (approved by NRC SERs, dated November 27, 1981 and September 4, 1984).

(g) Report, A. A. F. Ansari, K. J. Burns and D. K. Beller, "Methods for the Analysis of Boiling Water Reactors: Transient Critical Power Ratio Analysis (RETRAN-TCPYAOI)," YAEC-1299P, March 1982 (approved by NRC SER, dated September 15, 1982).

(h) Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (approved by NRC SER, dated November 30, 1977).

(i) Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A-9, GE Company Proprietary, September 1988, as amended.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

In addition to the changes requested by the licensee to implement the COLR in accordance with Generic Letter 88-16, the licensee also proposed a modification to the definition of the minimum critical power ratio (MCPR). The change to the definition is an administrative change and is, therefore, acceptable.

3.0 ENVIRONMENT CONSIDERATION

This amendment involves a change in the record keeping, reporting and administrative controls. The amendment also involves requirements with respect to installation or use of facility components located within the restricted areas, as defined in 10 CFR 20.3(a)(4). The staff has determined that the amendment involves no significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSIONS

We have reviewed the request by the Vermont Yankee Nuclear Power Corporation to modify the Technical Specifications of the Vermont Yankee plant that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the Specification. Based on this review, we conclude that these Technical Specification modifications are acceptable.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter (BVY-89-43) from Warren P. Murphy (VYNPC) to NRC, dated May 12, 1989.

2. Letter (BVY-89-67) from Warren P. Murphy (VYNPC) to NRC, dated July 14, 1989.

3. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: D. Fieno

Dated: September 15, 1989

AMENDMENT NO. 116 TO DPR-28 VERMONT YANKEE NUCLEAR POWER STATION DATED September 15, 1989

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